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Engineering and integration risks arising from advanced magnetic divertor configurations

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The divertor configuration defines the power exhaust capabilities of DEMO as one of the major key design parameters and sets a number of requirements on the tokamak layout, including port sizes, poloidal field coil positions, and size of toroidal field coils. It also requires a corresponding configuration of plasma-facing components (PFCs) and a remote handling scheme to be able to handle the cassettes and associated in-vessel components (IVC) the configuration requires.

There is a risk that the baseline ITER-like single-null (SN) divertor configuration cannot meet the PFC technology limits regarding power exhaust and first wall protection while achieving the target plasma performance requirements of DEMO or a future fusion power plant. Alternative magnetic configurations – double-null, snowflake, X-, and super-X – exist and potentially offer solutions to these risks and a route to achievable power handling in DEMO. But these options impose significant changes on machine architecture, increase the machine complexity and affect remote handling and plasma physics and so an integrated approach must be taken to assessing the feasibility of these options.

In this paper we describe the work being undertaken, and main results so far, in assessing the impact of incorporating these alternative configurations into DEMO whilst respecting requirements on remote handling access, forces on coils, plasma control and performance, etc.

Keywords: DEMO, systems studies, system modelling, fusion power plant, technology choices.

1. Introduction

The EUROfusion roadmap [1] targets the production of electricity from fusion in the 2050s. This implies that building work on DEMO must begin before 2040 with most final design decisions made well in advance to allow the completion of engineering design work. Technology choices must be made before this so that systems integration can be completed and requirements set. Overall, the conclusion is that in order to meet the roadmap target, we need technologies that exist today, at least in functional form; we cannot rely on breakthrough technologies that will arrive when required and slot into plant designs predicated on their existence [2].

In addition to the production of substantial output of electricity, DEMO aims to achieve tritium self-sufficiency, and also to demonstrate maintenance technologies and cycles consistent with the requirements of a commercial fusion power plant. Any structural design modifications to incorporate alternative technologies must also consider impacts on these targets. Finally, numerous studies into the economics of fusion power (e.g. [3]) have indicated the capital-intensive nature of fusion economics; that is, the initial investment in the plant is the largest driver of the final cost of electricity. One way of reducing the capital cost is to make the tokamak as small as possible for a target power output.

One of the principle size-drivers in a tokamak power plant is the performance of the divertor in terms of the power which can be allowed to cross the separatrix, much of which must be radiated away in order to achieve detachment and thus avoid significant erosion rates of the divertor surface [4]. The lowest-risk approach is to follow the path laid by ITER and make ITER-like assumptions for physics and technology performance, and this is the approach taken by the EUROfusion baseline design. However there is a risk that the baseline single-null (SN) divertor configuration cannot meet the plasma-facing component (PFC) technology limits regarding power exhaust and first wall protection while achieving the target plasma performance requirements of DEMO or a future fusion power plant. Alternative magnetic configurations – double-null, snowflake, X-, and super-X – exist and potentially offer solutions to these risks. In this first stage, impacts on magnet engineering and RM access are assessed, with more detailed modelling on remaining issues to follow on a downselected group of options.

2. DEMO baseline design

The current (2017) DEMO baseline design is summarized in Figure 1. It assumes modest advances on ITER physics and technology, with a target of 500MW net electrical power and a minimum pulse length of two hours. The divertor challenge quantifier, $P_{sep}B/qAR_0$, is

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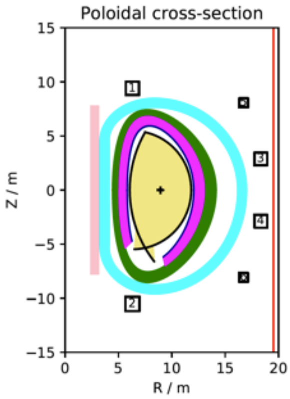
constructed by combining the Eich scaling [5] for scrape-off layer width with the tokamak geometry and conducted power loss and represents a measure of the power density on the divertor which is probably recoverable without significant damage should plasma detachment be lost [4]. For baseline DEMO, this value is scaled from ITER.



Characteristic	Value
R_0 / a (m)	8.9 / 2.9
κ_{95} / δ_{95}	1.65 / 0.33
Fusion power (MW)	2000
Burn time (s)	7200
$\beta_{N,tot}$	2.9
$P_{sep}B/qAR_0$ (MW T m^{-1})	9.2

Figure 1: Key parameters from the 2017 EU DEMO baseline. The final value is the divertor challenge quantifier.

3. Divertor options

3.1 Baseline single-null (SN)

This is the standard "ITER-like" divertor; a lower single-null of the type achievable on a wide range of existing experimental machines. For this work, it acts as the reference.

3.2 Double-null (DN)

An issue with the SN DEMO design is the power loading around the secondary X-point at the top of the machine, potentially requiring the use of limITERs. Such limITERs would occlude sections of the breeder blanket, reducing tritium production and electricity generation. Implementing a DN layout may avoid the power loading issues and allows access to potentially better-performing physics regimes, but would introduce additional remote maintenance (RM) complications (Figure 2).

In this case the inboard leg carries only a small fraction of the total divertor power [6] and the PFC could be incorporated into the inboard blanket segment. This option is currently being investigated within EUROfusion PPPT in Key Design Issue 1 (KDI1) [7]. The maintenance is then carried out by removal of a 'keystone' section containing the outer divertor section, which can be replaced independently. However, this requires additional RM operations, slowing component replacement procedures. An alternative configuration with midplane segmentation of the blankets takes advantage of the DN symmetry to reduce individual component mass; however this means that simultaneous divertor and blanket operations are no longer possible and requires very large spaces below the tokamak for access, further complicating building design and layout.

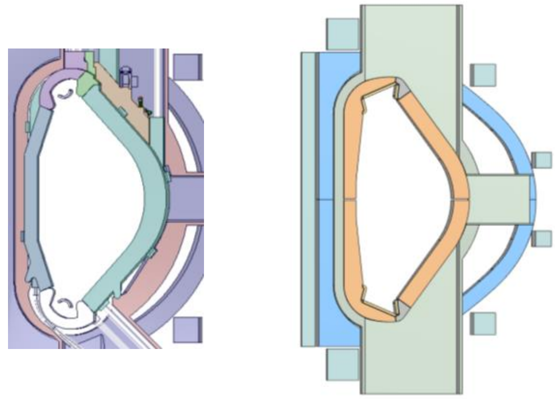


Figure 2: Possible double-null architectures. Vertical blanket access with lower divertor cassette (left) and split blanket access (right) [8].

3.3 Super-X (SX)

This configuration [9] aims to extend the strike point to high radius, and use a secondary null to spread the power over much larger areas (Figure 3).

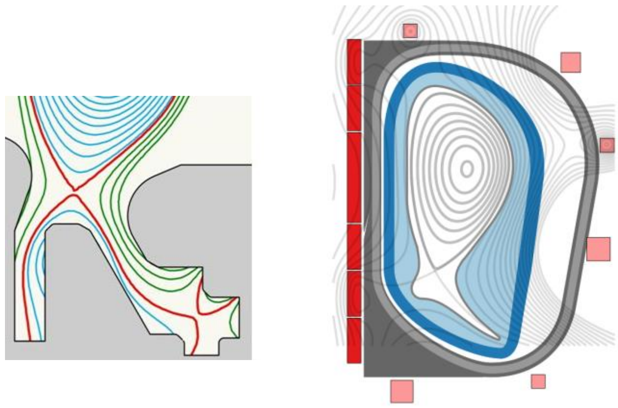


Figure 3: The basic super-X configuration (left) showing divertor leg extension and flux spreading, and an attempt to incorporate it into a DEMO-scale plasma without in-vessel coils (right).

However, some of the shortcomings of this approach in a real power plant rapidly become evident. The position of the PF coils mean that horizontal maintenance for the divertor segments is required – which makes aspects of the RM easier but means that blanket volume is lost to allow the access, reducing the tritium breeding performance (TBR) [10]. In addition, stress modelling indicates high out-of-plane loads acting on the TF coils challenging the design of the outer intercoil structures (Figure 4).

More problematically, from a plant-design perspective, the SX configuration only protects the outer divertor limb, in the case of a detached plasma. In SN, around 33% of the conducted power ends up on the inner limb and so a SN-SX is limited in the benefits that it can provide. The DN-SX would instead also lower the heat load in the inner strike point legs and thus robustly protect the divertor surfaces. For this reason, future studies will concentrate on the DN-SX configuration. However, without in-vessel coils, configurations providing the desired plasma geometry without hugely exceeding reasonable vertical force limits in the PF coils have proved elusive. The approach applied here is to start with a large array of virtual-coils and gradually eliminate them to identify a minimum set. Figure 4 shows one configuration with a minimum set of 10 PF coils, achieved before the forces in the coils exceeded 1000 MN per coil (already in excess of ready achievability). It was proposed that the maximum force limit is set to 400MN in DEMO, compared to the ITER value of 160MN [11]. No configuration was identified which would allow RM access while respecting force limits.

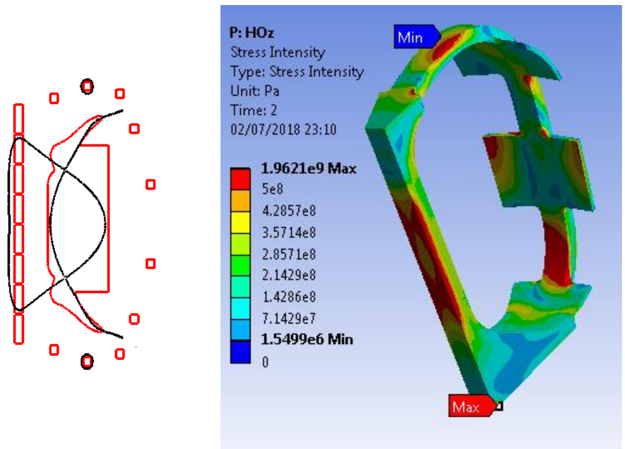


Figure 4: Magnet engineering issues for super-X. In the DN-SX variant (left) it was impossible to reduce the required number of PF coils to the level where RM access was achievable while keeping magnet forces below 1000 MN [12]. (Right) Locations of high through-casing stresses (red), between the outboard intercoil structures for SN-SX.

The principal path to avoiding these issues appear to be the use of in-vessel coils. Future work includes investigation of the feasibility of incorporating such coils, either copper-based or superconducting. Another alternative is reducing the outer limb length, and moving to an X-divertor (Section 3.5).

3.4 Snowflake (SF)

The SF configuration [10] induces a second magnetic null very close to the first, generating 4 divertor limbs 60° apart (Figure 5). However, although they have been demonstrated in a number of current machines, the physics of SF divertors remain underdeveloped and it is not clear how the power is shared between the limbs. On the other hand, it seems likely that the area around the X-point where the connection length is very long is large, allowing for high levels of X-point radiation which protect both inner and outer limbs. This permits a promising SN configuration which might allow for acceptable RM access to the divertor through a horizontal port (impact on TBR still to be assessed) and with reasonable forces in the PF coils. Previous work has shown that there is an impact on the flux swing supplied by the PF/CS coilset [6], but the global impacts of this can mitigated by improved divertor performance allowing lower radiative impurity levels in the plasma – if this improved performance can in fact be demonstrated. An additional impact is that the increased X-point radiation places high loads – up to 1MW m^{-2} -- on surfaces close to the X-point.

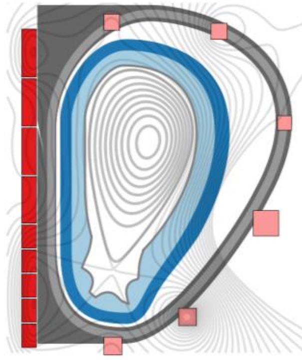


Figure 5: A snowflake equilibrium, showing the extra limb-splitting in the divertor. In this case the PF coil positioning optimization is not yet complete.

A more substantial concern is the control of the divertor heat loading under foreseeable plasma movements (Figure 6). Small perturbations to the plasma position can shift the divertor limbs – and the heat sharing between them – considerably. This may require defining the design heat load for each single target to be close to that for a standard SN configuration. Other effects are still to be studied. It may also be required to extend the high-heat-flux PFCs over the lower blanket segments, again impacting on TBR.

3D configurations are in preparation to investigate TF coil stresses, RM kinematics, and neutronics including TBR.

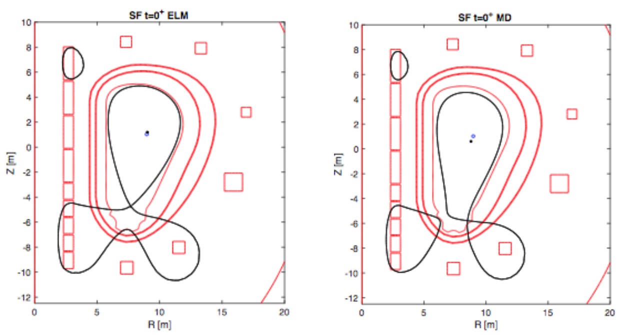


Figure 6: SF control issues – equilibria following an ELM (left) and the onset of a disruption (right).

3.5 X-divertor

The X-divertor resembles the SX but without the extended leg, saving space inside the TF coil and potentially reducing stresses, although also decreasing the divertor wetted area. Early SN configurations have been produced based on the SF coilset (Figure 7) but these are subject to the same limitations as the SN-SX configuration, namely that only the outer divertor limb is protected. DN configurations are in preparation, as are 3D TF coil configurations to investigate stresses.

4. Impacts on integration and design

4.1 Magnet design

It is clear that these alternative configurations pose additional challenges in TF coil design and PF configurations. In general they require more space and therefore the TF coils are larger, with higher stored energy and stresses, and the PF coils are further from the plasma and often in conflict with one another, requiring higher currents. With the exception of the DN configuration, the up-down asymmetry means the active power required to stabilize plasma perturbations is expected to be large [12]. The DEMO baseline TF coils are already borderline from a manufacturing perspective, and developing larger or more intricately shaped coils clearly increases DEMO project risks. These increased risks could be mitigated by the reduction of DEMO scope, or through reduction in overall device size which can be accomplished by more optimistic plasma physics assumptions: i.e. a risk transfer to reduced physics basis, for example higher performance regimes.

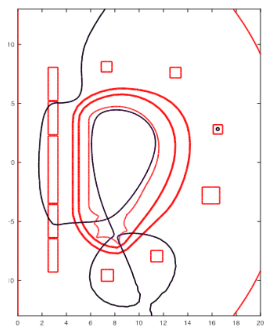


Figure 7: X-divertor configuration [11].

4.2 Remote handling

The PF coil layouts have been designed for the configurations outlined here with input from RM specialists, although kinematic studies and segmentation have not started yet for all configurations. Particular concerns relate to the size of divertor cassettes for the different configurations – particularly SX and SF – and the impact on other in-vessel components (IVC) from the horizontal access for these configurations, requiring a reduction in TBR as well as reconfiguration and repositioning of the ex-vessel systems to provide access.

4.3 Physics

All the alternative divertor configurations have a reduced physics basis over the ITER-like divertor, and therefore increased overall performance risks. Particular unknowns cover the actual stable radiative performance of SF divertors and their controllability with respect to plasma perturbations. In addition, it is unlikely that the required lines of sight for divertor diagnostics are available from the midplane ports in e.g. SX and SF configurations; these can be achieved but at the cost of additional vessel ports and IVC penetrations in the divertor region.

4.4 Other factors

The relative lengths of the TF coil can be used as a proxy for manufacturing costs, and these are shown in Table 1. As the TF coil length also provides a proxy for magnetized volume and hence stored magnetic energy, increasing length not only represents additional manufacturing cost but also increased safety considerations, particularly with regards to superconducting quench protection.

Concept	TF coil length (m)	Relative length / cost	Relative W
Baseline SN	48.5	1.00	1.00
DN	49.5	1.02	1.04
SX	52.7	1.09	1.18
SF	49.6	1.02	1.05

Table 1: TF coil lengths, relative costs, and relative stored magnetic energy W for the different concepts.

5. Conclusions

At this current conceptual design phase the direct costs of design changes are relatively small, except in programme delays as the alterations cascade through the integration process requiring analyses to be repeated and other systems to be modified. However some of these advanced configurations – in particular the Super-X – increase the execution risks of various systems in order to ease physics issues. While DEMO and any subsequent fusion power plant must consist of an integrated solution, possibly not provided by the ITER-like divertor, it is clear that there are no 'easy wins' offered by these configurations. With these caveats the current conclusions are:

1. **DN**: this configuration has a reasonably-developed physics basis and no show-stopping engineering issues identified yet, although is imposes considerable additional development on the RM systems with potential consequences for an impact on the plant availability.
2. **SX**: it has proven extremely challenging to find a configuration which respects RM access and magnet force limits whilst achieving an acceptable (DN) SX configuration. Further development will rely on speculative technology

such as in-vessel coils, rather than systems currently in development.

3. **X**: The work on these designs has not developed far enough yet to draw conclusions.
4. **SF**: Integration into a power plant design seems achievable, but the physics basis is still underdeveloped and hence possible performance is very difficult to quantify. Control issues may ultimately preclude integration.

Currently we have focused on magnets and RM access: further work, once 3D configurations are generated, will cover port configurations and IVC attachment and kinematics; exhaust pumping simulations in complex geometries; impacts on breeder blanket design including TBR; and assessment of plasma control issues. Finally capital cost and waste variations will be investigated.

Acknowledgments

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